Materials for DEMO and Reactor Applications - Boundary Conditions and New Concepts

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Abstract. DEMO is the name for the first stage prototype fusion reactor considered to be the next step after ITER towards realizing fusion. For the realization of fusion energy especially materials questions pose a significant challenge already today. Heat, particle and neutron loads are a significant problem to material lifetime when extrapolating to DEMO. For many of the issues faced advanced materials solution are under discussion or already under development. In particular components such as the first wall and the divertor of the reactor can benefit from introducing new approaches such as composites or new alloys into the discussion. Cracking, oxidation as well as fuel management are driving issues when deciding for new materials. Here W_f/W Composites as well as strengthened CuCrZr components together with oxidation resilient tungsten alloys allow the step towards a fusion reactor. In addition, neutron induced effects such as transmutation, embrittlement and after-heat and activation are essential. Therefore, when designing a component an approach taking into account all aspects is required.

1. Introduction and Boundary Conditions

When considering a future fusion power plant multiple intertwined issues need to be evaluated (fig. 1). Some of the main problems a future reactor is faced with are linked to the materials exposed to the fusion environment and their lifetime considerations

[1, 2]. Already from fig. 1 one can see that at the far branches of the tree multiple times the following issues arise: cooling media, neutron flux and neutron damage, ion impact and sputtering as well as heat loads and transient events.

In the following only a subset of those conditions can be evaluated and so far only for the relatively well known conditions of the next step devices such as ITER & DEMO [2].

1.1. DEMO Conditions

DEMO is presently considered to be the nearest-term reactor design that has the capability to produce electricity and is viewed in Europe [3] as a single step between ITER and a commercial power plant [4, 5, 6]. Currently, no conceptual design exists for DEMO apart from early studies. A design has not been formally selected, and detailed operational requirements are only now being developed [7], hence for discussion purposes we simply assume a reactor with the with fusion power of 2GW as given in [2, 8]. From the assumptions presented in [2] an average of typically $10-20\mathrm{MW/m}^2$ on the divertor is to be expected and with wall loading around $\sim 1-1.5MW$. For the neutron loading one can refer to [9] with (40dpa / 5fpy (full power year))

This machine is already significantly different in size and performance from the next step device, ITER. Main differences include significant power and hence neutron production , tritium self sufficiency, high availability and duty cycle as well as a pulse length of hours rather than minutes. In addition, safety regulation will be more stringent both for operation and also for maintainability and component exchange [7]. A reactor might even go beyond these requirements, e.g. steady state operation.

1.2. First Wall and Divertor

Several issues related to materials used in the construction of a future fusion reactor need still to be tackled. Among those are the issues related to the first wall and divertor surfaces, their power handling capabilities and lifetime. For the next generation device, ITER, a solution based on actively cooled tungsten (W) components has been developed for the divertor, while beryllium will be used on the first wall [10, 11]. The cooling medium will be water as is also considered for high heat load components in DEMO [7]. For the first wall of a fusion reactor unique challenges on materials in extreme environments require advanced features in areas ranging from mechanical strength to thermal properties. The main challenges include wall lifetime, erosion, fuel management and overall safety. For the lifetime of the wall material, considerations of thermal fatigue as well as transient heat loading are crucial as typically 10^9 (30Hz) thermal transients (ELMs) during one full power year of operation are to be expected. Tungsten (W) is currently the main candidate material for the first wall of a fusion reactor as it is resilient against erosion, has the highest melting point of any metal and shows rather benign behavior under neutron irradiation, as well as low tritium retention. Erosion of the first wall and the divertor will in addition require a significant armor thickness or short exchange intervals, while high-power transients need strong mitigation efficiency to prevent damage of the PFCs. [11].

For the next step devices, e.g. DEMO, or a future fusion reactor the limits on power exhaust, availability and lifetime are quite stringent. As conventional mono-blocks are allowing for $10MW/m^2$ [11] and transmutation and radiation damage can quickly diminish the thermal conductivity to 50% [12] .Radiation effects including neutron embrittlement may hence limit actively cooled W components in DEMO to about 3-5 MW/m^2 due to the diminished thermal conductivity or the need to replace

CuCrZr with Steels with its low thermal conductivity [13, 8]. Quite extensive studies and materials programs [14, 15, 16, 1] have already been performed hence it is assumed that the boundary conditions [8] be fulfilled for the materials are in many cases above the technical feasibility limits as they are understood today.

- Extended power handling, i.e., ability to withstand power loads larger than $10\,\mathrm{MW/m^2}$. Here especially the choice of coolant is critical. Water cooling might be required to allow sufficient exhaust at given acceptable pumping power [2, 8].
- The radiation damage for the divertor is predicted to be close to 3 dpa/fpy. For copper if chosen the value varies between 3 and 5 dpa / fpy
- It is assumed that despite the radiation damage erosion of the armor is the dominant lifetime determining factor. Here it needs to be considered that maxium thickness is also determined by the required neutron transmission required for tritium breeding.
- Even when starting up DEMO in phases a final blanket could be requred to withstand up to 50 dpa in order to minimize the exchange frequency.

In the following we will however try to concentrate on three groups of issues [8, 7]

- Power exhaust and energy production: The first wall blanket exhausts the neutron power and hence must be operated at elevated temperatures to allow for efficient energy conversion. Here a material must be chosen with a suitable operational temperature window and sufficient exhaust capability. The cooling medium for high temperature operation can be crucial.
- Mitigate the effect of material degradation due to neutrons and reduce radioactive waste: One can select materials that allow high temperature operation, mitigate effect of operational degradation such as embrittlement and neutron effects linked to transmutation.

- Tritium self-sufficiency and safety: 22 kg/year of tritium are required for a 2GW plasma operated at 20% availability, this means $\sim$ 85% [8] of the in-vessel surface must be covered by a breeding blanket and the loss of tritium without ability to recover needs to be minimized.
- Accident scenarios need to be considered e.g. loss of coolant and air ingress are to be considered.

2. Material Issues

As an example the divertor lifetime is considered as the leading parameter. Fig. 2(a) depicts what typically is seen as the main avenues of damage to the material of the divertor. Either high heat-loads cause melting, cracking or recrystallization or neutrons impact the actual microstructure of the material. Surfaces are damage by impacting ions causing both surface morphology changes and erosion.

Fig. 2(b) depicts hence one approach to solve at least some of the problems. Choosing tungsten (W) as the main armor material suppresses sputtering due to the high atomic mass compared to the sputtering ions. Tungsten also has a rather high thermal conductivity (Values at RT; W: $\sim$ 173W/(mK) , Cu: $\sim$ 390W/(mK), steel: $\sim 17W/(mK)$) and can hence facilitate higher heat exhaust than e.g. steel. For tungsten also the high melting point is beneficial. Thermal properties, however are intrinsically linked to potential transmutation and irradiation processes (sec. 2.4). In addition, tungsten has a rather low hydrogen solubility and hence facilitates low retention under fusion conditions [17, 18]. Tungsten is, however, inherently brittle and does show catastrophic oxidation behavior at elevated temperatures.

2.1. Operational Window

Based on the assumption that W so far is the option to be used as the armor layer of the reactor PFCs already quite basic assumptions can be made when picking the operational window and thickness of such components.

The lower operating temperature limit in metals and alloys is mainly determined by radiation embrittlement (decrease in fracture toughness), which is generally most pronounced for irradiation temperatures below $\sim 0.3\,T_{melt}$, where T_{melt} is the melting temperature (tungsten $\sim$ $3600K$) [19]. The upper operating temperature limit is determined by one of four factors, all of which become more pronounced with increasing exposure time such as thermal creep (grain boundary sliding or matrix diffusional creep), high temperature helium embrittlement of grain boundaries, cavity swelling (particularly important for Cu alloys), and coolant compatibility such as corrosion issues.

This is depicted in fig. 3(b) with fig. 3(a) showing the thus given operational conditions for a given cooling structure. Fig. 3(b) also indicates with arrow the direction in which new or advanced materials should extend the operational windows.

If the PFC surface is operated around 800 K inside the operational window for W, and copper is chosen together with water as part of the coolant solution the thickness (d_i) is automatically determined (with κ_i the heat conductivity). In a simplified one-dimensional approach for two materials (1,2) one can write:

$$q = \frac{T_{surface} - T_{cool}}{d_1/\kappa_1 + d_2/\kappa_2} \tag{1}$$

This means that the maximum heat exhaust is determined by the heat conduction, the potential for recrystallization and the ductile-to-brittle transition behavior of the material. For a real component this simple approximation will not hold and temperature gradients along the surface will be present - causing additional thermal stress and inhomogeneous changes in material properties. Here new material options are required to allow a larger operational window, by overcoming brittleness issues, keeping in mind that a maximized heat conduction is crucial (e.g. steel).

For transient events the limits can even be more stringent when considering the limited penetration depth of a given heat pulse fig. 3(c) and its maximum surface temperature rise (eqn. (2)) with κ the heat conductivity, ρ the density and c the heat capacity).

$$\Delta T^{\infty}_{surface}(t) = \frac{q_s}{\sqrt{\kappa\rho c}\cdot\sqrt{\pi}}\sqrt{\Delta t} \tag{2}$$

Active cooling for fast transients is not relevant because of the small penetration depth. From assumptions related to unmitigated ELMs at 1 GW/m^2 for 1ms [11] already a temperature rise of 1500K is achieved within only the top 1 mm. Despite the ELMs being poloidally distributed along the target an unmitigated ELM can still deposited 1 GW/m^2 locally. Additions along the poloidal directions will only aggravate the problem.

Cracking or melting is difficult to prevent under such conditions , a loss of control might push close to the limit as the top surface at 10-20 MW/m^2 will already operate close to 2000 $°C$ [2] . Irreparable damage has to be avoided. Fig. 3(d) depicts even higher thermal wall loads caused by so called disruptions, sudden and uncontrolled loss of the plasma with deposition of the energy on the wall. Assuming that 50% of the thermal energy is radiated during the thermal quench of the plasma and with a limited toroidal and poloidal inhomogeneity of two respectively the thermal disruption loads are always much above the crack limit of W [20] although still below the melt limit. Variations of the torus geometry (aspect ratio) provides only a moderate reduction of the thermal loads [21].

2.2. Evolution of Thermal Properties

In addition to the above mentioned issues fig. 4(a) shows that the fusion environment can also drastically change some of initial material parameters. Already a low amount of transmutation can have a significant influence on the power-exhaust. When calculating the thermal conductivity based on $\kappa \cdot \rho = L \cdot T$ with κ the thermal conductivity, ρ the resistivity and L the Lorenz number with a value of $3.2 \times 10^{-8} W \Omega K^{-2}$ for tungsten one can estimate that κ drops by 60% already at 5 wt% of Re or Os. Already insignificant transmutation irradiation can change the thermal properties of W. From fig.4(b) one can determine that especially at lower temperatures κ drops significantly. In any case stable and predictable material properties are necessary even under radiation - or a detailed knowledge of the time dependent evolution, to determine lifetime and performance of components.

2.3. Embrittlement

Conventional high performance materials offer high strength and stiffness combined with low density hence weight. However, a fundamental limitation is the inherent brittleness of tungsten. As seen above cracking hence brittle behavior can be a limiting factor when operating any tungsten based PFCs in a tokamak [20]. For the fusion environment the additional problem results from operational embrittlement.

Fig. 5(a) shows that already at a moderate neutron fluence corresponding to 1 dpa the DBTT of tungsten moves up to almost 630K. If in addition recyrstallisation occures (fig. 5) almost no structural load can be given to the tungsten component at temperatures of a few hundred degrees. For a typical mono-block [11, 22] a tungsten thickness of 6mm on top of the CuCrZr cooling pipe would mean, based on simple estimations (eqn. 1) that only the top part of a exposed mono-block would be in the allowed temperature range specified in fig. 3(b). This means for a water-cooled solution tungsten is normally brittle hence only a functional part, suppressing e.g. erosion and allowing for high operational temperatures. Failure is usually sudden and catastrophic, with no significant damage or warning and little residual load-bearing capacity if any. Structures that satisfy a visual inspection may fail suddenly at loads much lower than expected. Cracking must usually be avoided for PFCs and certainly for structural components.

2.4. Activation and Transmutation

An issue that can be quite crucial, especially for complex components with multiple material and alloying components, is the activation and subsequent recyclability under neutron irradiation. As fusion is typically considered a technology with minimal or no long term nuclear waste [23] tungsten and special steel grades [24] have optimized radiation performance with respect to low activation. Molybdenum and aluminium are avoided as they produce long term activation products [9, 23]

Fig. 6(a) shows the activation behavior for various elements under a typical fusion neutron exposure with a duration of five years for materials exposed at the first wall. Based on a study provided in [6, 9] with a neutron flux at the first wall of $1.0 \times 10^{15}\mathrm{ncm}^{-2}\mathrm{s}^{-1}$. For materials exposed in the divertor a factor 10 lower neutron rate is expected in the area of the high heat flux exposure due to geometrical reasons [7].

Fig. 6(b) shows the values of an assumed component containing W, Cr, Cu and erbium, representing e.g. a mono-block with small interlayers and a copper cooling structure. Already here it is clear that the shielded hands on radiation level can not be achieved after 100 years when using copper in cooling structures at the first wall. Mitigation of these effects need to be considered by utilizing non or low activation materials. e.g. replacing copper for the first wall and removing Er or Al oxides in favor of yttria.

2.5. Retention and Permeation

Tritium retention in PFCs due to plasma-wall interactions is one of the most critical safety issues for ITER and future fusion devices. For carbon-based PFCs the co-deposition of fuel with re-deposited carbon has been identified as the main retention mechanism (fig. 7). This retention grows linearly with particle fluence and can reach such large amounts that carbon is omitted in the activated phase of ITER and therefore basically excluded for future reactors due its large issues related to retention [17] . Instead, tungsten is foreseen as PFC material in the divertor of ITER and tungsten based alloys are the most promising candidates for PFCs in future reactors. Fuel retention behavior of tungsten is subject to present studies. It was shown that by replacing CFC with W in the Joint European Torus (JET) the retention e.g. can be significantly reduced [18]. An issue that however remains is the potential for diffusion of hydrogen into the material. In the breeding blankets especially the interaction of tritium with reduced activation ferritic martensitic (RAFM) steels (e.g. EUROFER-97) can be crucial to minimize fuel retention or loss.

3. New Material Options

For all the above determined issues or boundary conditions potential solutions need to be developed. We are faced with a multilayer approach for the PFCs including armor, fuel barriers, cooling structures & breeding elements and hence we have to consider a multitude of interfacing materials. From the plasma towards the cooling structure we consider tungsten or tungsten alloys on either copper or steel based structures with functional layers, e.g. permeation barriers or compliance layers. A generally new components concept to circumvent classical definitions of limits is required, applying damage resilient materials such as composites, followed by a much better definition what can be tolerated before a component needs to be exchanged. We need to define lifetime for PFCs with more parameters than erosion and cracking. Composite approaches to enhance material parameters and mitigate damage modes by utilizing mixed properties will be ideal including safety features like passivating alloys etc. Not yet developed ideas on self-healing or damage tolerant materials similar to aerospace applications can be a future field of research, including e.g. liquid metals [25]. Already today smart materials, fibre composites and alloys which adapt to the operational scenario are possible. In some cases detrimental effects such as erosion are actually used to facilitate material functions (sec. 3.2). If W as a first wall material is required to suppress erosion even preferential sputtering can turn the top layer of alloys or steel into a thin layer of erosion suppressing tungsten [26, 27, 28].

3.1. Composites for High Loads

The basic idea is to introduce extrinsic mechanisms which allow energy dissipation. This is the only way to enhance the toughness in brittle materials [29]. A basic strategy to achieve pseudo-ductility is the incorporation of fibres and a weak interface into a matrix, which needs extensive development and validation [30]. To overcome brittleness issues when using W, a W fibre enhanced W composite material (W_f/W), incorporating extrinsic toughening mechanisms can be used. The extrinsic mechanisms enable energy dissipation and thus stress peaks can be released at crack tips and cracks can be stopped. Another option are composite laminates made of commercially available raw materials [31, 16]. The link between W_f/W and laminates is the similarity of fibres and foils. Both show a special microstructure of highly deformed and elongated grains, hence showing high strength and ductility even at room temperature [32, 33, 34]. Accordingly, even in the brittle regime, below the DBTT, theses materials allow for a certain tolerance towards cracking and damage in general. In comparison, conventional tungsten would fail immediately. From fig. 8(a) the principle of fibre-composite strengthening behavior can be seen. Even when a crack has been initiated inside the material the energy dissipation mechanisms allow further load to be put towards the component. After reaching the ultimate strength other mechanisms lead to a controlled failure rather than a catastrophic one in the brittle case. First W_{f}/W samples have been produced, showing extrinsic toughening mechanisms similar to those of ceramic materials [35, 36]. These mechanisms will also help to mitigate effects of operational embrittlement due to neutrons and high operational temperatures. A component based on $\mathrm{W_f/W}$ can be developed with both chemical vapor deposition (CVD), utilizing a CVD setup, and a powder metallurgical path through hot isostatic pressing [37, 35]. Crucial in both cases is the interface between fibre and matrix. The interface is a thin layer (fig. 8(b)) with targeted properties: weak enough to enable the toughening mechanism, as strong as possible to maximize the dissipated energy [38]. This is an idea based on enabling pseudo-ductile fracture in inherently brittle material e.g. SiC ceramics [39].

Keeping in mind the above mentioned boundary conditions one can consider that brittleness from either neutron irradiation or elevated temperatures can be mitigated as the pseudo-ductilisation does not rely on any part of the material being ductile, crack resilience can be established [35, 36]. Facilities to produce both CVD as well as powder metallurgical $\mathrm{W_f/W}$ are readily available.

In order to enable the use of composites in fusion, it needs to be shown that for new materials equally good behavior in terms of thermal conductivity, erosion and retention can be established. As part of the development particularty the choice of the fibre and interface material can be crucial. A sag-stabilized potassium doped fibre can even retain some ductility in addition to strengthening the material [40, 34]. For the fibre-matrix interface a non activating choice is required hence one should move from the so far considered erbia [38, 35] potentially towards the low activating yttria.

In addition to conventional composites also fine grain tungsten is an option to strengthen and ductilize tungsten [32] similar to other metals [33] . An option to achieve this for W is powder injection molding (PIM) [41, 42]. PIM as production method enables the mass fabrication of low cost, high performance components with complex geometries. The range in dimensions of the produced parts reach from a micro-gearwheel (d=3 mm, 0.050 g) up to a heavy plate ((60x60x20)mm, 1400 g). Furthermore, PIM as special process allows the joining of tungsten and doped tungsten materials without brazing and the development of composite and prototype materials, as described in [41]. Therefore, it is an ideal tool for divertor R&D as well as material science. Figure 9(a) show new developed tungsten parts produced via PIM for a study of plasma-wall interaction at ASDEX Upgrade at IPP Garching. Uniaxial grain orientation (see fig. 9(b)), up- & down scaling, good thermal shock resistance, shape complexity and high final density are several typical properties of PIM tungsten materials. Detrimental mechanical properties, like ductility and strength, are tunable in a wide range (example: W-1TiC and W-2Y2O3) [42]. Based on these properties the PIM process will enable the further development and assessment of new custom-made tungsten materials as well as allow further scientific investigations on prototype materials.

3.2. Tungsten Smart Alloys

Addressing the safety issue, a loss-of-coolant accident in a fusion reactor could lead to a temperature rise of the first wall components of 1400 K after approximately $30-60$ days due to neutron induced after heat of the in-vessel components [6] as schematically depicted in fig. 10(a).

Thereby, a potential problem with the use of W in a fusion reactor is the formation of radioactive and highly volatile tungsten oxide (WO_3) compounds. In order to suppress the release of W oxides tungsten-based alloys containing vitrifying components seem feasible, as they can be processed to thick protective coatings with reasonable thermal conductivity, e.g. by plasma spraying with subsequent densification as already demonstrated for titanium and tantalum coatings [43]. Enhanced erosion of light elements during normal reactor operation is not expected to of concern. Preferential sputtering of alloying elements leads to rapid depletion of the first atomic layers and leaves a pure W surface facing the plasma as per the given different sputtering yields. [44, 45]. This mechanism is similar to the above mentioned EUROFER-97 surface enrichment. Fig. 10(b) displays the basic mechanism. During operation plasma ions erode the light constituents of the alloy, leaving behind a thin depleted zone with only tungsten remaining. Subsequently, the tungsten layer suppresses further erosion, hence utilizing the beneficial properties of tungsten. In case of a loss-of-coolant and air or water ingress the tungsten layer oxides releasing a minimum amount of WO_3 and then passivating the alloy due to the chromium content. W-Cr-Y with a tungsten fraction of up to 70 at% shows a 10^4-fold suppression of tungsten oxidation due to self-passivation [46]. Test systems are being produced via magnetron sputtering and evaluated with respect to their oxidation behavior. Production of bulk samples is ongoing. Rigorous testing of oxidation behavior, high heat flux testing and plasma loads as well as mass production for candidate materials are under preparation. The material can be considered for both first wall and divertor applications especially when combined with the strengthening properties of the W_f/W composite approach. The PWI behavior and potential neutron or temperature embrittlement still needs to be assessed.

3.3. Functionally Graded Materials

Having discussed tungsten as the main candidate for the PFMs of a fusion reactor the joint to the underlying cooling structure or wall structure in general is crucial. From the differing thermal expansion coefficients for the different materials (copper $\sim 16.5\mu$m/(mK), tungsten: $\sim 4.5\mu$m/(mK) , stainless steel: $\sim 12\mu$m/(mK)) it is clear that a mature solution of joining them needs to be established.

As example systems the development of functionally graded materials (FGMs) between W as the PFM with the structural material EUROFER-97 can be considered, exhibiting complementary volumetric gradients of W and EUROFER-97. As depicted in [47] FGMs are promising candidates for interlayers between components of two different materials especially when considering applications such as the blanket modules of a DEMO [7, 48] or even a helium cooled tungsten divertor with low to medium heat-flux (1-5MW/m^2) for which the heat conductivity of EUROFER-97 may be sufficient. Fig. 11 shows a potential development cycle, which comprises further optimization of FGM manufacturing techniques, joining, mechanical testing and thermal cycling. This will determine the viability of the FGM concept and also allow the comparison with conventional joints. Similar ideas are developed for the transition between copper and W [49, 50] potentially being used as solution for a water-cooled high heat-flux divertor [7, 8]

3.4. Permeation Barriers

Moving from the plasma facing material towards the structural part of the reactor tritium management is an issue in particular for the breeding blankets. In order to prevent fuel loss and radiological hazards it is important to suppress permeation of tritium towards the cooling channels. Research on tritium permeation barriers ranges over a variety of materials [51, 52, 53, 54, 55, 56, 57], including alumina and erbia. Permeation barriers require high permeation reduction factors, high thermal stability and corrosion resistance as well as similar thermal expansion coefficients compared to those of the substrate. Investigation of the permeation reduction factor requires controlled experiments. A new gas-driven permeation setup is established at FZ Jülich to investigate deuterium permeation e.g. through different ceramic coatings on EUROFER-97, which significantly reduce the deuterium permeation. Several deposition techniques can be considered, e.g. filtered arc deposition, chemical routes, and magnetron sputter deposition. A reduction factor of 50-100 is essential to allow a safe operation and a reasonable tritium breeding ratio. In addition to permeation reduction and mechanical feasibility, compatibility with neutron irradiation needs to be considered. Here especially the promising barrier candidates alumina and erbia do have issues. Yttria has a better activation behavior as those candidates, see fig. 6. First permeation measurements of yttria coatings on EUROFER-97 show a similar permeation reduction factor as erbia [58]. Studies on yttria are ongoing.

4. Summary and Outlook

Considering all the above mentioned issues when using materials in a fusion reactor environment a highly integrated approach is required. The lifetime of PFCs and joints due to erosion, creep, thermal cycling, and embrittlement needs to be compatible with steady state operation and short maintenance intervals. Thermal properties of composites and components have to be at least similar to bulk materials when enhanced properties in terms of strength are not to hinder the maximization of operational performance. Damage resilient materials can here facilitate small, thin components and hence higher exhaust capabilities. The components need to be compatible with the aim of tritium breeding and self-sufficiency and hence mitigate tritium retention and loss.

Despite using various alloying components, interlayers or coatings maintainability and recycling of used materials is required to make fusion viable and publicly acceptable. Last but not least, large scale production of advanced materials is crucial. We hence propose to utilize the composite approach together with alloying concepts to maximize the potential of the tungsten part of a potential PFC. Together with W/Cu composites at the coolant level and W/EUROFER-97 joints high-performance components can be developed. Rigorous testing with respect to PWI and high heat-flux performance are planned for all concepts to have prototype components available within 5 years for application in existing fusion devices.

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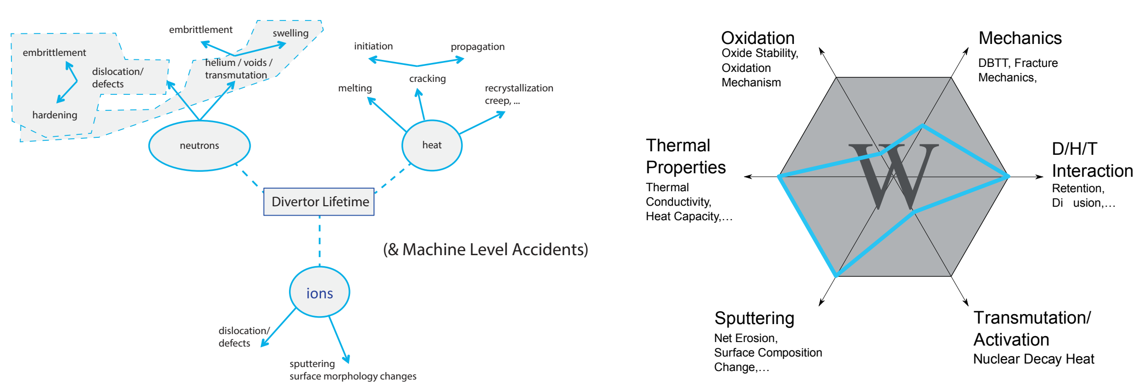
Figures

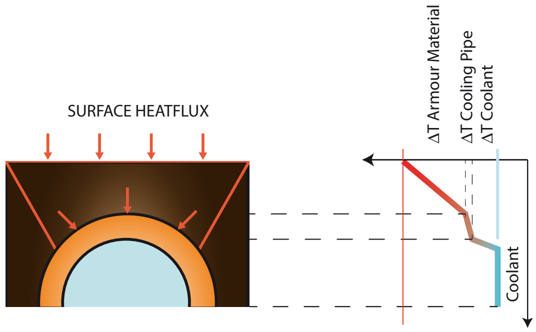
Figure 1: Materials in Fusion face not a single but a multitude of interlinked challenges.



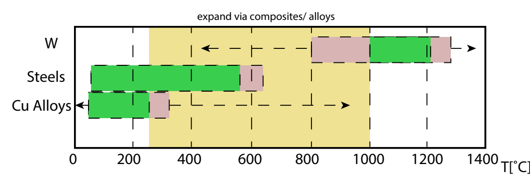
(a) Materials Related Issues

(b) Tungsten as a material in fusion

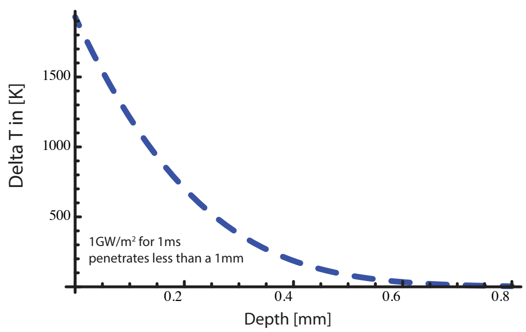
Figure 2: Material Issues for PFCs.



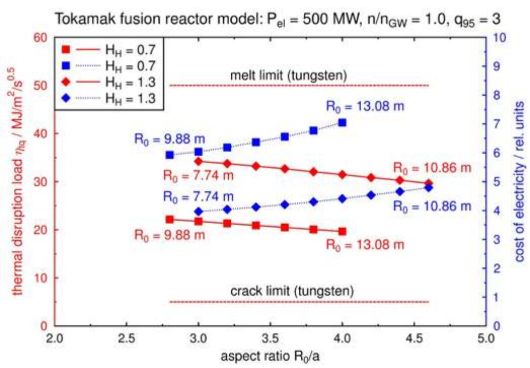
(a) Steady state heat flux in a conventional monoblock like structure



(b) Operational Windows for Structural Materials in Fusion (based on [59])

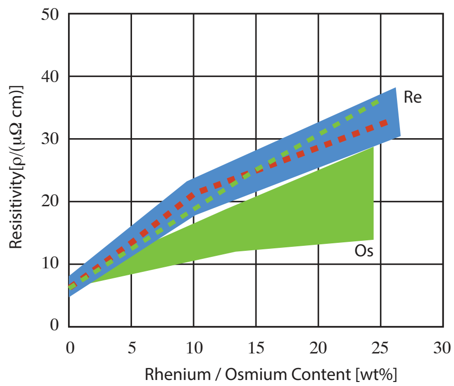


(c) [Heat flux penetration (in tungsten) from $1GW/m^2$, after 1ms

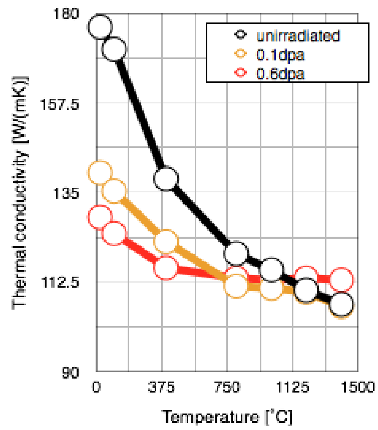


(d) Disruption heat loads - material limits [21, 11, 20]

Figure 3: Power-exhaust - Issues arising from steady state and transients

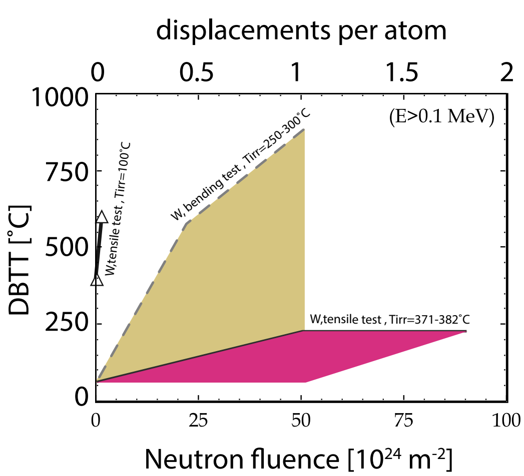


(a) Electrical resistivity of W containing various amounts of Re or Os. The blue band shows WxRe, the green band depicts WxOs for various cases irradiated up to 1.5 dpa. The red line and green line stand for W-xRe and W-xOs unirradiated respectively [12], $T_{irr} = 300 - 750°C$

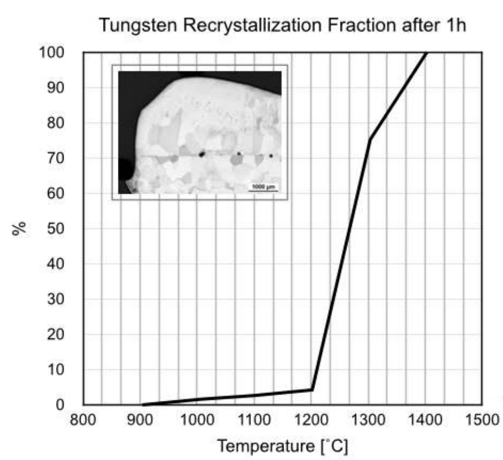


(b) Thermal conductivity of W before and after neutron irradiation (0.1 and 0.6 dpa, $T_{irr} = 200°C$) [22]

Figure 4: Change of electrical and thermal properties of tungsten under neutron irradiation and transmutation

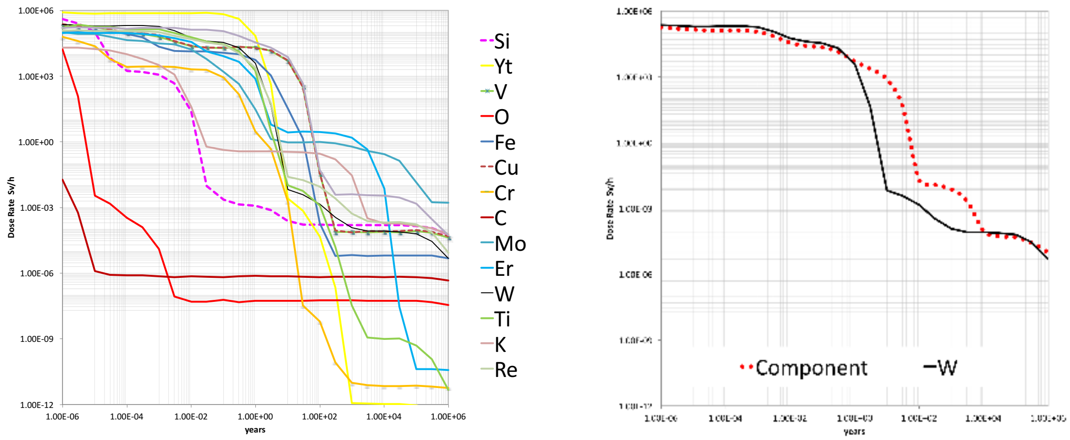


(a) DBTT dependence after neutron irradiation based on [23]



(b) Recrystallization of tungsten after 1h at given T, based on [?]

Figure 5: Factors determining operational embrittlement are neutron irradiation and recrystallization



(a) Residual activation of various elements

(b) W 79.7wt%, Er 0.6wt%, Cr 12.1wt%, Cu 7.5%

Figure 6: The activation of materials for the first wall can be estimated as an upper bound (based on [9]). Divertor components in general are less prone to activation. Shielded hands-on level: $2\, mSv/h$, Hands-On Level: $10\, \mu Sv/h$

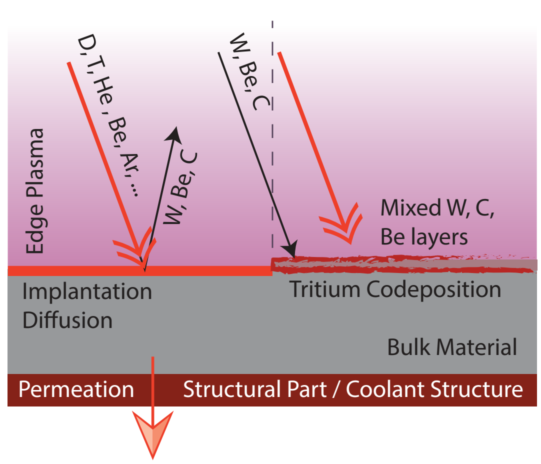
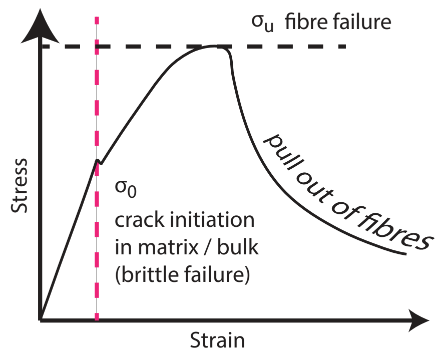
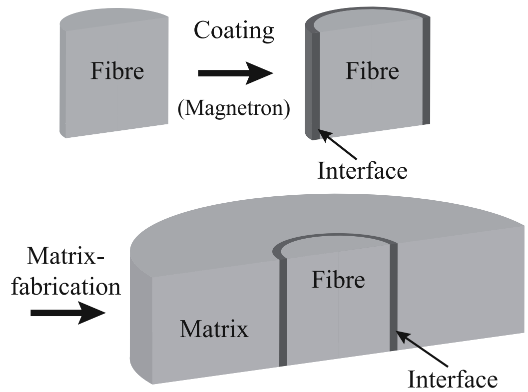


Figure 7: Fuel retetention and permeation issues under plasma exposure conditions



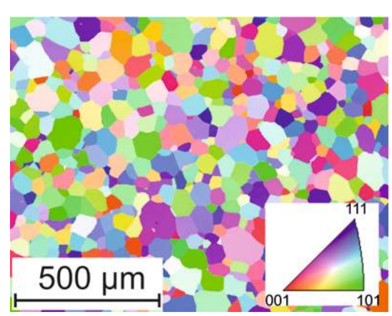
(a) Strain Stress Curve for a typical Composite Material



(b) fibre enhanced composite and interface layer

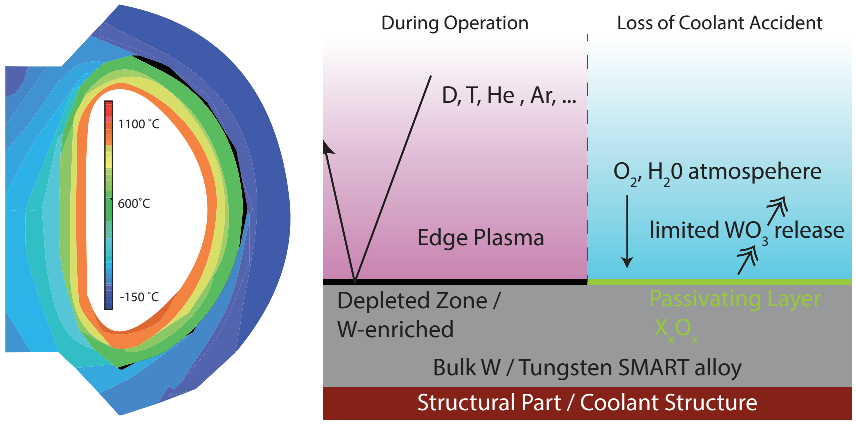
Figure 8: Composite approaches based on pseudo ductilisation.

(a)



(b)

Figure 9: W-PIM parts produced for a study of plasma-wall interaction in ASDEX Upgrade (a), EBSD image of the uniaxial grain orientation (b).



(a) Loss of coolant accident scenarios - Wall temperature estimate based on [6], 10 days after incident

(b) Working principle of a smart alloys based PFC with both the operational and accident mechanisms shown.

Figure 10

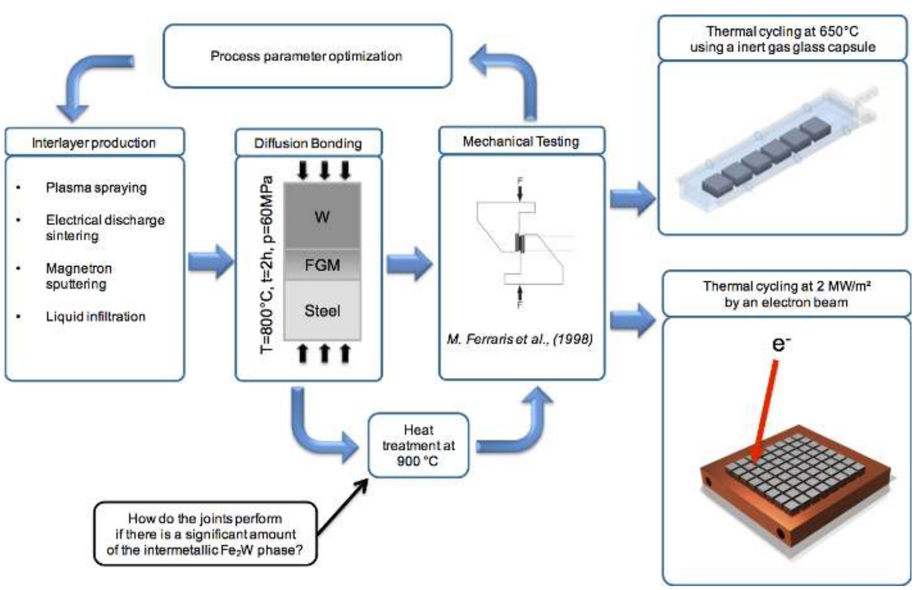


Figure 11: Functionally Graded Materials and a potential optimization and testing procedure.